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Low-Enriched Fuel Design Concept for the Prismatic Very High Temperature Reactor Core

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Abstract – *A new non-TRISO fuel and clad design concept is proposed for the prismatic, helium-cooled Very High Temperature Reactor core. The new concept could substantially reduce the current 10-20 wt% TRISO uranium enrichments down to 4-6 wt% for both initial and reload cores. The proposed fuel form would be a high-temperature, high-density uranium ceramic, for example UO_2, configured into very small diameter cylindrical rods. The small diameter fuel rods significantly increase core reactivity through improved neutron moderation and fuel lumping. Although a high-temperature clad system for the concept remains to be developed, recent success in tube fabrication and preliminary irradiation testing of silicon carbide (SiC) cladding for light water reactor applications offers good potential for this application, and for future development of other carbide clad designs. A high-temperature ceramic fuel, together with a high-temperature clad material, could also lead to higher thermal safety margins during both normal and transient reactor conditions relative to TRISO fuel. The calculated neutronic results show that the low-enrichment, small diameter fuel rods and low thermal neutron absorbing clad retain the strong negative Doppler fuel temperature coefficient of reactivity that ensures inherent safe operation of the VHTR, and depletion studies demonstrate that an 18-month power cycle can be achieved with the lower enrichment fuel.*

I. INTRODUCTION

The prismatic Very High Temperature Reactor (VHTR) is the leading Generation IV reactor concept destined for near-term deployment in the United States. This reactor concept is a thermal spectrum reactor characterized by a relatively low core power density (6.6 W/cm^3), annular active core composed of prismatic fuel blocks, and an inner and outer graphite reflector. The low power density, large graphite mass, and strong negative Doppler fuel coefficient of reactivity ensures that this reactor core concept will adhere to the underlying Generation IV safety goal of inherent safety under all accident or transient conditions.[1] In addition, the high outlet gas temperature (900–950 °C) allows for efficient electricity generation (48% Brayton cycle) and nuclear heat and hydrogen production. Together these attributes make the prismatic VHTR the number one Generation IV contender for near-term deployment in the U.S.

There are currently two prismatic VHTR design variants differentiated primarily by coolant and core size/ power rating. The first is a gas-cooled reactor that uses high pressure helium gas and has a core power rating of approximately 600 MW$_{th}$. The second is a liquid salt-cooled core[2] with a much larger 2400-4000 MW$_{th}$ core power rating. The design analysis herein, as applied to the new fuel and clad design concept, is based on the gas-cooled VHTR at 600 MW$_{th}$ and specifically on the General Atomics Modular Helium Reactor concept.[3] However, both the helium-cooled and the liquid salt-cooled reactor concepts could accommodate and benefit from the new design concept, since both VHTR variants anticipate the use of graphite prismatic fuel blocks.

In addition to the use of a prismatic graphite fuel block, both the helium- and liquid salt-cooled VHTRs will use TRISO-coated particle fuel.[4] The TRISO particle offers a high-integrity, high-pressure boundary for the containment of fission product gases. The TRISO particle itself, however, with its relatively small uranium kernel volume, multiple coatings, and relatively low particle packing fraction in the fuel compacts does not allow for efficient loading of uranium in the fuel rods. Also, the relatively large diameter fuel compacts displace the higher density block graphite which negatively impacts the neutron moderation and the self-shielding of U238. These TRISO fuel characteristics together with the current particle packing fraction limitation of <35% in the fuel compacts further limit the TRISO uranium block loadings and results in a significant negative reactivity penalty. The new fuel and clad design concept described herein will attempt to remedy this situation.

The major drawback of the new fuel and clad design concept at present is the lack of a proven clad system. However, recent investigations[5] into silicon carbide clad performance for light water reactors, silicon carbide tube manufacturing methods, and ongoing irradiation tests may be the beginning of a new clad development era for both light water and high-temperature VHTRs as well.

II. EXISTING TRISO FUEL AND CORE DESIGN

The following five sections briefly describe: (1) the prismatic VHTR reactor core, (2) the basic prismatic fuel block design, (3) the current TRISO particle fuel design, (4) TRISO fuel reactivity characteristics, and (5) the required fuel block uranium loadings using TRISO particle fuel to achieve an 18-month power cycle. These introductory sections are intended to illustrate the reactivity problems associated with the current TRISO fuel block designs and set the stage for the new low-enrichment fuel and clad design.

II.A. General Atomics GT-MHR

The General Atomics Gas Turbine-Modular Helium Reactor[3] or GT-MHR was selected as the basis for our prismatic block VHTR neutronic evaluations. The GT-MHR core contains both prismatic fuel and reflector blocks. The fuel blocks are arranged in an annular active core in hexagonal rings 6, 7, and 8 for a total of 102 fuel columns (Figure 1). Each active fuel column is a vertical stack of ten fuel blocks (1,020 total core fuel blocks).

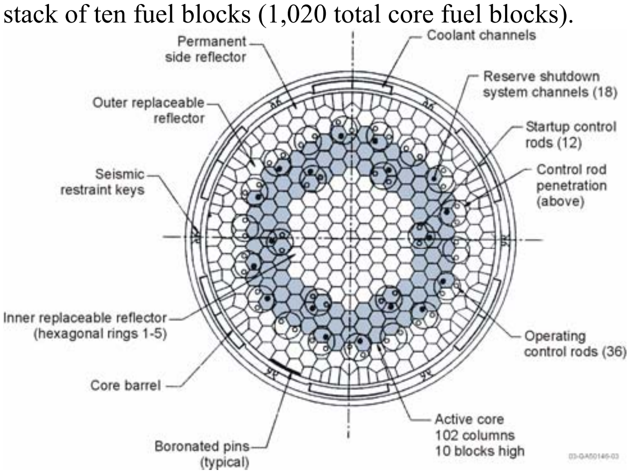

Figure 1. The General Atomics GT-MHR reactor core.

Although the GT-MHR has three different types of fuel blocks in the core: standard, reserve shutdown, and control rod, we will assume for simplicity that all fuel blocks in our models and analyses are standard fuel blocks. The standard fuel block is described next.

II.B. Fort Saint Vrain Fuel Block

The GT-MHR standard fuel block is basically the same design as the old hexagonal Fort St. Vrain (FSV) standard fuel block shown in Figure 2. A standard FSV fuel block has a flat-to-flat dimension of approximately 35.82 cm and a height of 79.3 cm. The block graphite has a density of 1.74 g/cm^3. Fuel and coolant channels are drilled axially into the block and arranged on a triangular lattice pitch (1.88 cm). Each coolant channel is surrounded by six fuel rod channels. In each standard FSV fuel block there are 108 coolant channels and 210 fuel channels. The fuel channels are smaller in diameter (12.7 mm) than the coolant channels (16.0 mm) and are distinguishable in Figures 2 and 3.

Figure 2. The standard FSV graphite fuel block.

II.C. TRISO Particle Fuel

The fuel block channels are to be filled with fuel compacts. These are cylindrical pellets, 12.45 mm in diameter and 49.3 mm in length. A typical fuel rod in a prismatic block would contain 15 vertically stacked compacts with a total fuel rod length of approximately 74 cm. Each fuel compact would contain TRISO-coated fuel particles bound in a graphite matrix. Figure 3 shows a more detailed cross sectional view of standard GT-MHR and VHTR fuel block for TRISO fuel.

The basic TRISO-coated particle consists of a central spheroid kernel (350 μm diameter) of uranium oxy-carbide ($UC_{0.5}O_{1.5}$) or uranium oxide (UO_2) coated with multiple layers of carbide materials. The first coating around the kernel is a relatively thick (100 μm), low density (1.0 g/cm^3) graphite buffer to absorb fission fragment kinetic energy and accommodate fission product gases and semi-volatile species. The buffer layer is coated with a 35 μm thick, high-density (1.9 g/cm^3) pyrolytic graphite layer or inner pyrolytic coating (IPyC). The next coating is a 35 μm thick SiC layer (3.2 g/cm^3) designed to contain fission product migration and to provide high-strength pressure vessel containment for the particle. The final coating is 40 μm thick, high-density (1.87 g/cm^3) pyrolytic graphite layer known as the outer pyrolytic coating (OPyC).

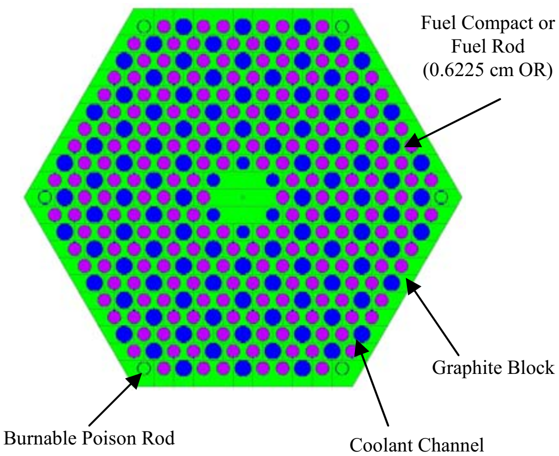

Figure 3. The standard GT-MHR and VHTR TRISO fuel block design.

Currently, the TRISO-coated particles are manufactured with a 350 μm UCO kernel diameter and a total particle diameter of approximately 0.770 mm. A new TRISO particle with a larger diameter UCO kernel (425 μm) and a total particle diameter of 0.845 mm has been proposed[6] to allow heavier uranium block loadings for reload blocks and to alleviate compact packing fractions from reaching the 35% maximum fabrication limit.

Reactor physics design parameters associated with the TRISO-coated particle and the fuel compacts include: (1) kernel diameter, (2) kernel density, (3) uranium enrichment, (4) particle packing fraction (PF), and (5) compact diameter or fuel rod radius. These parameters can be adjusted to achieve desired fuel block uranium loadings and reactivity. Although parameters 1-4 can all be used to increase the fuel block uranium load, only an increase in the uranium enrichment and/or a reduction in fuel rod radius can increase the core reactivity. Since our goal is to reduce the current enrichment levels of 10 wt% for the initial core[6] and 14-20 wt% for a 2-batch reload fuel block[6], our only choice for increasing the core reactivity is to reduce the fuel rod radius.

II.D. Reactivity and Fuel Rod Radius

The prismatic VHTR core reactivity can be substantially increased if the fuel rod radius is reduced to a value less than the current 0.6225 cm radius dimension. To demonstrate the potential reactivity gain, consider the following calculated lattice k-infinity curves as a function of fuel rod radius shown in Figure 4. In this example, the following parameters are fixed: the uranium enrichment (10.36 wt%), UCO kernel size (350 μm), and kernel density (10.5 g/cm^3). The curves are plotted parametrically as a function of the particle packing fraction.

The lattice k-infinity increases substantially as the fuel rod radius decreases from 0.6225 cm, for all packing fractions. The increase in k-infinity, or reactivity, is attributed to a more favorable carbon-to-uranium (C:U) lattice ratio and the lumping of the fuel to increase the U238 self-shielding or resonance escape probability. The fuel rod radius reduction produces both a reduction in fuel rod uranium and an increase in the lattice carbon. In the latter case, low density compact matrix graphite (1.1 g/cm^3) is replaced with higher density (1.74 g/cm^3) bulk graphite from the block. The fuel lattice is subsequently transformed from an under-moderated lattice with a C:U ratio of approximately 362 into a more reactive lattice with a more optimal C:U ratio in the range of 1300–1500 for fuel rod radii in the 0.2–0.4 cm range. This is the basis for the smaller fuel rod radius that will play a central role in the new fuel and clad design.

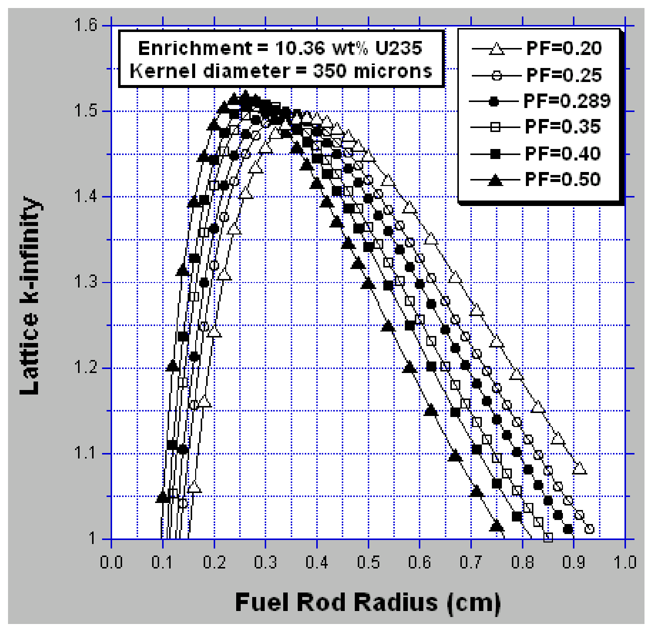

Figure 4. Lattice k-infinity versus TRISO fuel rod radius.

It should be noted that a reduction in the TRISO fuel rod radius decreases the amount of uranium in the fuel rod and block, and subsequently reduces the core reactivity and power cycle length. The uranium block loading can readily be increased however, by increasing the particle packing fraction, kernel diameter, or kernel density, or all three. Unfortunately, an increase in any one or all of these parameters results in a further decrease in core reactivity. Again, only an increase in the uranium enrichment will actually boost the TRISO core reactivity, and consequently is the primary reason for the relatively high uranium enrichment required for the prismatic VHTR core.

Curves similar to those in Figure 3 are also obtained for TRISO particle enrichments in the range of 10-20 wt% and kernel sizes ranging from 350-425 μm. The curves are simply shifted up (higher reactivity) for higher enrichments.

II.E. Uranium Loading

For the TRISO VHTR initial core, a uniform fuel block mass loading of 554 g U235 per block with an enrichment of 10.0 wt% is required to achieve the Gen IV power cycle length goal of 18 months (540 days). This mass loading requires a particle packing fraction of 0.247 for TRISO particles with the larger 425 μm UCO kernel. It should be noted that even this larger UCO kernel accounts for just 12.7% of the total TRISO particle volume, and only 3.14% of the total fuel compact volume, demonstrating the inefficient uranium loading of TRISO fuel particles in the fuel rods.

Similarly, for a 2-batch TRISO reload block, a uranium mass loading of approximately 776 g U235 per block with a minimum enrichment of 14% is required to achieve the 18-month cycle length. This mass loading also requires a particle packing fraction of 0.247 for TRISO particles with the 425 μm UCO kernel. Again, the UCO kernel accounts for just 12.7% of the total particle volume and only 3.14% of the total compact or fuel rod volume.

The relatively small UCO kernel volume and the limited particle packing fraction (<0.35) limits the uranium mass density loading and hence requires a relatively high enrichment and relatively large fuel rod diameter (12.45 mm) in order to accommodate enough TRISO-coated particles to meet fissile uranium cycle requirements. The relatively large TRISO fuel rod not only displaces the high density block graphite, but also reduces the lumping or self-shielding of the U238, further impacting the core reactivity.

III. NEW FUEL AND CLAD DESIGN CONCEPT

The new fuel and clad design concept presented next is intended to improve the prismatic VHTR core reactivity by taking advantage of the small diameter rod effect. In order to use a small diameter rod, a non-TRISO, high-uranium density or solid solution fuel form must be used to achieve sufficient uranium block loading for power cycle requirements. Also, the fuel form should be a high-temperature ceramic material with a high melting point in order to achieve a large fuel thermal safety margin under transient conditions.

It is assumed for now that the new fuel and clad concept will utilize the same basic FSV fuel block with the same number of coolant, fuel, and burnable poison channels. A more optimal block design with different flat-to-flat and length dimensions, number of fuel and coolant channels, and fuel rod pitch, is reserved for future work.

III.A. Small Diameter Ceramic Fuel Rods

Fuel rods with diameters smaller than the current TRISO 12.45 mm diameter rods will require a high-uranium density fuel form. Table I lists four potential candidate fuel forms for the new fuel concept along with their melting points and density.

TABLE I

Potential candidate fuel forms for small diameter fuel rods.

Fuel Form	Melting Point (ºC)	Density (g/cm^3)
UO_2	2800	10.96
UO_2-ZrO_2-CaO	2670	6-7
UN	2630	14.31
UC_2	2350-2400	11.28

Note that the melting points of these fuel forms are all considerably higher than the 1600 ºC temperature limitation[4] of the TRISO particle fuel. Above this temperature, the SiC coating in the TRISO particle begins to decompose.

The four fuel forms in Table I also have relatively high fissile uranium densities that will be useful for the new small fuel rod diameter concept to create a highly reactive fuel lattice. These fuel forms would be fabricated into cylindrical pellets or rods with optimal diameters in the range of 2-4 mm; rod diameters much smaller than the current GT-MHR and FSV 12.45-mm TRISO fuel compact diameters. Table I is not intended to be a complete list of all possible fuel forms, but rather to show potential and currently available fuel forms that can meet our design requirements.

Of the four fuel forms listed in Table I, UO_2 has been chosen here as the baseline fuel form to demonstrate the neutronic properties of the new fuel and clad design concept in the VHTR core. Low-enriched UO_2 is used in every commercial U.S. light water reactor today and has an extensive fuel performance database. In addition, fuel fabrication techniques for UO_2 pellets are mature and well-known. Consequently, the expectation is that the current UO_2 experience would greatly benefit the design of a VHTR with a low-enriched UO_2 fuel and present a minimal design risk.

The other three fuel forms in Table I are not as widely used. The UO_2-ZrO_2-CaO fuel form has a lower density and thermal conductivity than UO_2 and for these reasons alone may not be as attractive as UO_2. The other two high-temperature fuel forms, UN and UC_2, have densities greater than UO_2 and could possibly be used to further reduce the fuel rod diameters, and further increase core reactivity. Additional neutronic analysis and material research would be required to select the best fuel form.

III.B. Reactivity and Fuel Rod Radius

In order to estimate an optimal UO_2 fuel rod diameter for the prismatic VHTR core, a 1/12-core model was used to calculate the core k-effective as a function of the UO_2 fuel rod radius and uranium enrichment in the 4-6 wt% range. The core model is described in section IV.D and the k-effective curves are shown in Figure 5.

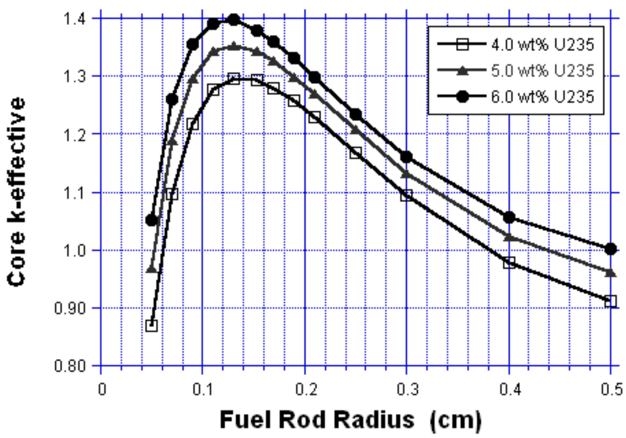

Figure 5. Core k-effective versus UO_2 fuel rod radius and enrichment.

Relative to the calculated TRISO infinite lattice k-infinity curves (Figure 4), the UO_2 core k-effective curves here are similar in shape and exhibit a maximum reactive region. In Figure 5, as the fuel rod radius decreases from 0.5 cm, the k-effective curve increases to a maximum value. The maximum k-effective values occur for fuel rod radii in the range of 0.12–0.16 cm. Therefore, the optimal UO_2 fuel rod diameters would range from 0.24–0.32 cm for the 4-6 wt% enrichment range. However, in order to stay on the under-moderated side of the curves, the fuel rod diameter should be ≥ 0.30 cm, or radius ≥ 0.15 cm.

III.C. UO_2 versus TRISO Uranium Loading

This section provides a comparison of fuel rod characteristics for TRISO particle and UO_2 loaded fuel blocks. Table II presents data for TRISO and UO_2 initial core fuel blocks, each with a 554 g U235 per block load, or enough to allow the initial core to achieve an 18-month power cycle.

TABLE II

Initial core fuel comparison for TRISO and UO_2.

Characteristic	TRISO	UO_2
Enrichment (wt%)	10.0	5.0
Fuel Rod Dia. (mm)	12.45	3.06
Fuel Rod Vol. (cm^3)	94.68	5.76
Fuel Density (g/cm^3)	10.5	10.4
U235 Block Mass (g)	554	554
U235 Rod Mass (g)	2.64	2.64
Fuel Volume Fraction (%)	3.14[a]	100
U235 Rod Density (g/cm^3)	0.028 [b]	0.458

a. 425 µm UCO kernel particle, packing fraction = 0.24715
b. Homogenized over fuel rod volume

Of particular interest are the large differences in (1) the enrichment (10 versus 5 wt%), (2) fuel rod diameter (12.45 versus 3.06 mm), (3) fuel rod volume (94.68 versus 5.76 cm^3), and (4) U235 rod density (0.028 versus 0.458 g/cm^3).

Table III presents data for TRISO and UO_2 reload fuel blocks, each with a 776 g U235 per block load, or enough to allow the 2-batch reload core to achieve an 18-month power cycle.

TABLE III

Reload core fuel comparison for TRISO and UO_2.

Characteristic	TRISO	UO_2
Enrichment (wt%)	14.0	5.0
Fuel Rod Dia. (mm)	12.45	3.63
Fuel Rod Vol. (cm^3)	94.68	8.06
Fuel Density (g/cm^3)	10.5	10.4
U235 Block Mass (g)	776	776
U235 Rod Mass (g)	3.70	3.70
Fuel Volume Fraction (%)	3.14[a]	100
U235 Rod Density (g/cm^3)	0.039 [b]	0.458

a. 425 µm UCO kernel particle, packing fraction = 0.24715
b. Homogenized over fuel rod volume

Of particular interest again are the large differences in (1) the enrichment (14 versus 5 wt%), (2) fuel rod diameter (12.45 versus 3.63 mm), (3) fuel rod volume (94.68 versus 8.06 cm^3), and (4) U235 rod density (0.039 versus 0.458 g/cm^3).

It is clear that the UO_2 fuel rods can easily pack the same amount of U235 into each fuel rod as the TRISO rod, but with a much smaller diameter fuel rod. In addition, small increases in the UO_2 fuel rod radius leads to relatively large increase in the uranium mass in the block. This is a very desirable feature of the UO_2 prismatic fuel block. Also, the small diameter UO_2 fuel rods can provide substantial self-shielding of the U238 and reactivity advantage.

III.D. UO_2 Small Diameter Fuel Rod Fabrication

Small diameter uranium oxide (UO_2) and mixed oxide (MOX) fuel rods have previously been fabricated for the Hanford Fast Flux Test Reactor (FFTF) fuel. A fuel manufacturing line at Hanford Engineering Development Laboratory was set up to manufacture high quality precision diameter oxide fuel pellets. The pellets were cylindrical fuel rods on the order of 0.4 mm in diameter (Figure 6), a diameter approximately 5-10 times smaller than the optimal fuel rod diameters proposed. The manufacture of small diameter UO_2 fuel rods for the VHTR in the 2-4 mm range should therefore be readily feasible.

Figure 6. FFTF stainless steel fuel pin containing 0.4 mm diameter UO_2 fuel pellets.

III.E. Clad Material and Design

The high-temperature clad material and design for this concept is the major unknown. No practical or proven high-temperature clad system yet exists for our application here. However, recent and ongoing research and development activities related to fabrication and irradiation testing of silicon carbide (SiC) clad systems for commercial light water[5] and Generation IV supercritical water reactor applications shows promise. The radiation resistance and strength provided by SiC tubes may eventually develop into a viable clad system for the VHTR as well. The SiC development could also provide the impetus for development of other carbide materials for future advanced clad system designs.

Much research and development will be required to achieve the desired high performance clad for VHTR application. However, selection of a low thermal neutron absorption clad material is not difficult and several will be presented for neutronic considerations. The neutronically acceptable clad material will also need to act as a pressure containment vessel for fission gases and be compatible with fuel and graphite block during irradiation. Important issues such as clad pressurization, gas release, pellet-clad-graphite block interaction, thermal conductivity, creep, oxidation, radiation damage, and strength, all at relatively high VHTR temperatures, must also be addressed.

Use of the proposed small diameter UO_2 fuel rods has a feature that may be useful in the final clad system design. There is space available to accommodate a relatively thick clad with thicknesses ranging perhaps from 1–4 mm. Such thicknesses would allow for some flexibility in clad system design.

Potential clad system designs for the small diameter UO_2 fuel rods/pellets might include a single thick sheath with a gas plenum similar to a standard light water reactor fuel rod, or a more complex system with multiple concentric sheaths. For example, a duplex SiC/SiC system[5] consisting of an inner monolithic SiC sheath with an outer sheath of SiC fibers wound around the inner sheath and then infiltrated with vapor to form a SiC matrix. The inner monolithic sheath contains the fuel and provides a hermetic seal for fission gases, while the outer composite sheath protects the inner sheath and adds overall strength. Other design features might include a low density graphite layer inside the inner SiC sheath to act as a buffer to minimize fuel-clad interaction and improve fission product absorption and retention, similar to the TRISO low-density buffer coating around the UCO kernel. There are many possible clad design options that could be explored.

Table IV lists some potential candidate clad materials along with their melting points and densities. Other materials may exist, but these materials have relatively low thermal neutron absorption cross sections.

TABLE IV

Potential high-temperature clad materials

Clad Material	Melting Point (ºC)	Density (g/cm^3)
ZrC	3540	6.73
ZrN	2980	7.09
SiC	1600-2200[a]	2.7-3.2
NbC	3608	7.82

a. Recent bulk SiC structures survived sintering temperatures >2200 °C

Of the Table IV candidates, zirconium carbide (ZrC) would be the most attractive clad material from the standpoint of its high melting point (3540 °C) and relatively low neutron absorption cross section. Silicon carbide is also a good candidate and may be able to withstand temperatures in excess of 2200 °C. Silicon also has a slightly lower thermal neutron absorption cross section relative to zirconium.

Selection of the best clad material would require a material research and development program that would include a sheath fabrication program and an irradiation test program. Clad system design would necessarily have to be a part of future research and development work and would represent the greatest effort and risk for the new fuel and clad design concept presented here.

Ultimately, the clad UO_2 fuel rods would be inserted into the prismatic block fuel channels. The fuel rods could be either the full length (74 cm) or shorter length clad rods stacked vertically in a fuel channel.

III.F. UO_2 and TRISO Block Design Comparisons

Figure 7 shows the new UO_2 fuel block design as it might look with the small diameter fuel rods (0.15 cm) and cladding (0.15 cm thickness) in a standard FSV prismatic fuel block. This figure is a cross sectional view of just a portion of a fuel block. If one compares the current GT-MHR or VHTR prismatic fuel block loaded with TRISO particle fuel compacts (Figure 3) with the UO_2 fuel block in Figure 7, the smaller diameter UO_2 fuel rods are apparent, as is the greater amount of block graphite.

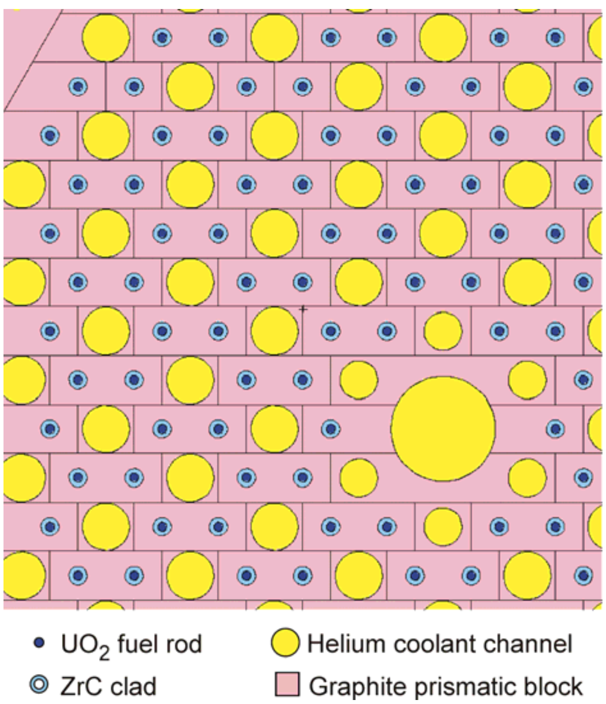

Figure 7. The new UO_2 fuel and clad prismatic block design concept.

IV. COMPUTER CODES AND MODELS

This section briefly describes the three reactor physics computer codes (MCNP5, ORIGEN2.2, and MOCUP) used in the neutronic analyses that follow in section V. This section also includes a description of the MCNP5 core model.

IV.A MCNP5

The MCNP (Monte Carlo N-Particle) code[7] Version 5, or MCNP5, is a general purpose, continuous energy, generalized geometry, coupled neutron-photon-electron Monte Carlo transport computer code. The powerful geometry capability allows for fully-explicit, three-dimensional cell representation of all components in a nuclear reactor core. The code can be used to calculate many different reactor physics parameters that include: cell neutron fluxes, neutron spectra, nuclear reaction rates, fission powers, and core eigenvalues. Continuous-energy neutron cross sections are employed by the code and typically range from 10^{-10} to 20 MeV for a wide variety of nuclides; the photon cross section energy range is from 1 keV to 100 MeV. Cross section data libraries are available from the Evaluated Nuclear Data Files version 5 and 6.

IV.B ORIGEN2.2

The ORIGEN2.2 (Oak Ridge Isotope Generation) Version 2.2 code[8] is used to calculate the complex time-dependent and coupled behavior of both radioactive and stable isotopes under constant power or flux conditions. Isotope production and destruction mechanisms include transmutation or neutron radiative capture, fission, threshold particle reactions, and radioactive decay processes. The code mathematical basis uses the matrix exponential method to solve large numbers of coupled ordinary differential equations relating isotopic concentrations with a high degree of accuracy. The ORIGEN2.2 code is used here as part of the depletion calculations.

IV.C MOCUP

The MOCUP (MCNP-ORIGEN2 Coupled Utility Program) code[9] is a system of external processors that links input and output files from the MCNP5 and ORIGEN2.2 codes in order to perform time-dependent depletion calculations. MOCUP performs specific, sequential tasks during each burnup iteration or timestep. No modifications are required to the MCNP5 or ORIGEN2.2 codes in order to run the MOCUP code system.

IV.D MCNP Core Model

An explicit, three-dimensional, 1/12-core MCNP5 model was specifically developed for the neutronic and depletion analyses (Figure 8). Reflective surface boundary conditions are applied to the two azimuthal surfaces (wedge sides) and the top and bottom surfaces to create an infinite axial extent and azimuthally symmetric core. The model includes solid graphite blocks for the inner and outer reflector regions; some outer reflector blocks have axial holes for control rods. All active core fuel blocks are assumed to be standard fuel blocks

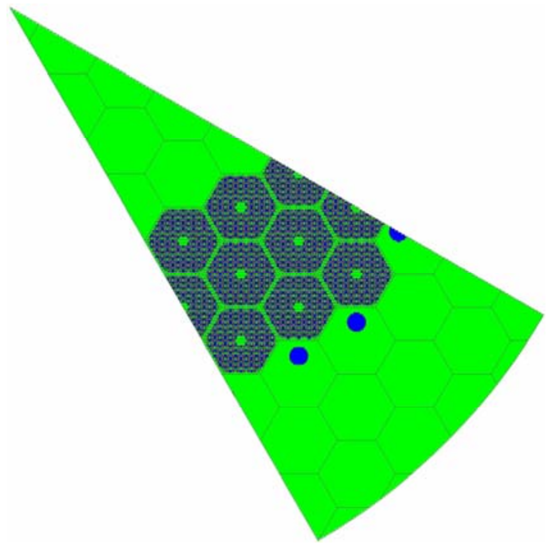

Figure 8. MCNP5 1/12-core model.

For the TRISO VHTR core models, the fuel compacts are assumed to be homogenized; TRISO particle double heterogeneity is not accounted for in the models. For the new fuel and clad design concept, the UO_2 pellets or fuel rods are assumed to be clad in SiC (0.15 cm thick). In both the TRISO and UO_2 1/12-core VHTR models, the fuel rod pitch is fixed at 1.8796 cm. The fuel temperature is assumed to be 1,000˚C. The structural graphite temperature in the fuel block, and inner and outer graphite reflector blocks, is assumed to 927˚C. The helium gas pressure is assumed to 7.12 MPa.

V. NEUTRONIC ANALYSES AND RESULTS

This section focuses on the neutronic behavior of a VHTR core with small diameter UO_2 fuel rods relative to a comparable TRISO VHTR core. Calculated neutronic results are presented for the Doppler fuel coefficient of reactivity, moderator coefficient of reactivity, neutron spectra, and core depletion comparisons for initial and reload fuel blocks of TRISO and UO_2 fuels.

V.A Doppler Fuel Coefficient

The Doppler fuel coefficient of reactivity for the UO_2 VHTR core at beginning-of-life was calculated using the 1/12-core model with the following assumptions. The UO_2 uranium enrichment was 5.0 wt% and the UO_2 fuel rods had a diameter, length, and pitch of 0.1534 cm, 73.95 cm, and 1.8796 cm, respectively. Each fuel block had a 554.26 g U235 per block initial core loading.

The calculated core k-effective as a function of fuel temperature (uranium temperature) is shown in Figure 9 over the range of 0–1600 °C. The core k-effective decreases with increasing fuel temperature and is indicative of a negative Doppler coefficient of reactivity.

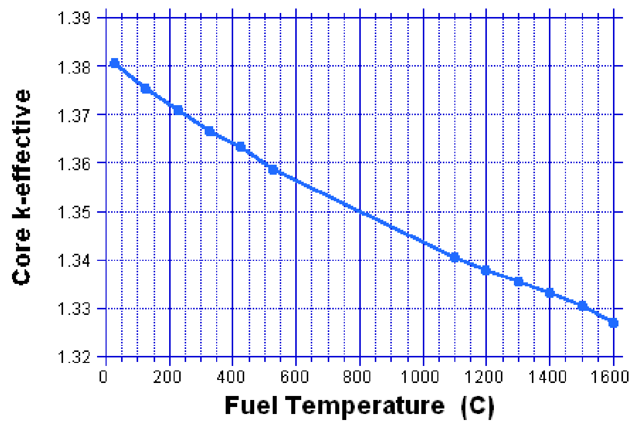

Figure 9. Core k-effective versus fuel temperature.

The corresponding Doppler fuel temperature coefficient of reactivity ranges from -0.052 to -0.025 mk/°C over the 0–1600 ºC temperature range. Although the coefficient decreases with increasing fuel temperature (factor of 2), it remains strongly negative and is the main reason for the VHTR inherent safety feature. These UO_2 Doppler coefficients are comparable in magnitude and behavior to those previously calculated for the TRISO particle VHTR core with fuel blocks of similar uranium loading.[10]

V.B Moderator Temperature Coefficient

The calculated core k-effective as a function of the bulk graphite or moderator temperature is shown in Figure 10 over the range of 0–1700 ºC.

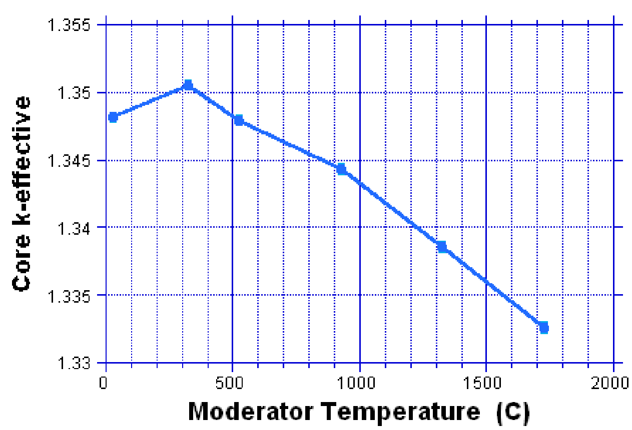

Figure 10. Core k-effective versus fuel temperature.

The core k-effective increases slightly over the low end of the temperature range (0-300 ºC), indicative of a small, but positive moderator temperature coefficient (MTC) of reactivity, and then decreases with temperatures above 300 ºC, indicative of a negative MTC.

The magnitude of the MTC achieves a maximum value of approximately +0.008 mk/°C over the 0–300 ºC temperature range, and then becomes negative and relatively constant over the 300–1700 ºC temperature range with a value range of -0.010 to -0.015 mk/°C. These moderator temperature coefficients for the UO_2 VHTR core are again comparable in magnitude and behavior to those previously calculated[5] for the TRISO particle VHTR core.

V.C Neutron Spectra Comparison

A comparison of the neutron energy spectra for the 5.0 wt% UO_2 and the 14 wt% enrichment TRISO particle fuel are presented in Figure 11. Note that the small diameter UO_2 fuel rod thermal neutron peak is higher and shifted slightly downward in energy. The softer UO_2 spectrum is attributed to the more optimal C:U ratio and better U238 self-shielding relative to the TRISO fuel rod lattice.

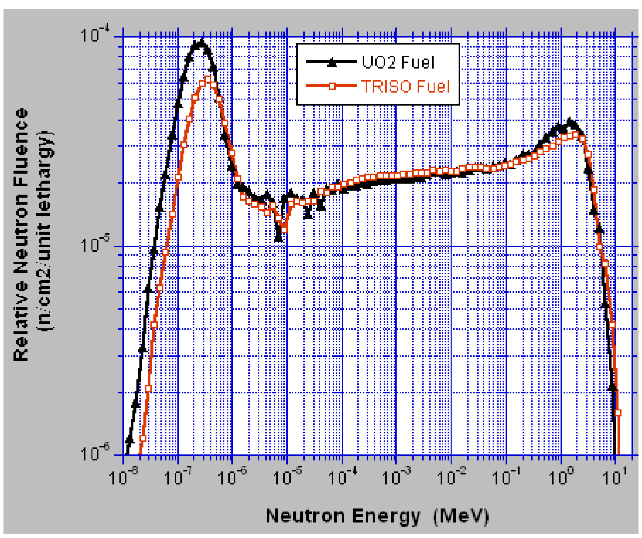

Figure 11. Neutron spectra comparison for the small diameter UO_2 and standard TRISO fuel rods.

V.D Core Depletion Analyses

Three depletion calculations were performed to compare and demonstrate the burnup capability of the proposed UO_2 small diameter fuel and clad concept relative to the TRISO-coated fuel. The three depletion calculations were all performed using the 1/12-core VHTR model and the MCNP5/ORIGEN2.2/MOCUP computer codes. Results from previous depletion calculations[6] for the TRISO VHTR core are used for comparison purposes here.

For the depletion comparisons, two core configurations are considered. The first core configuration is assumed to be a core uniformly loaded with fresh fuel blocks containing 554 g U235 per block. This U235 block loading and core configuration would be typical of the initial core for the TRISO VHTR and an 18-month power cycle length.

The second core configuration is hypothetical in that the entire active core is assumed to be uniformly loaded with fuel blocks containing 776 g U235 per block. This block loading is typical of a reload block in a 2-batch core reload scheme for the TRISO VHTR core where only half the core would be replaced with these reload blocks. For a more one-to-one depletion comparison however, the depletion calculations here assumed a uniform and full load of these 776 g reload blocks, and hence one would expect a 50% increase of the 18-month power cycle before the core goes subcritical (k-effective <1.0).

The first depletion calculation was the initial core with all active core fuel blocks loaded with 554 g U235 per block. For the TRISO fuel, the enrichment was 10.0 wt% U-235 with a particle packing fraction of 0.24715, UCO kernel size of 425 µm in diameter, UCO density of 10.50 g/cm^3, and a fuel rod diameter of 12.45 mm. For the UO_2 case, the enrichment was only 5.0 wt% with a fuel rod diameter of 3.06-mm. The reactivity letdown curves (Figure 12) show the TRISO fueled core goes subcritical at approximately 570 EFPD (Effective Full Power Day at 600 MW_{th} total core power) and the UO_2 core at 630 EFPD. The UO_2 core achieves an extra 60 EFPDs or an 11% increase over the TRISO core.

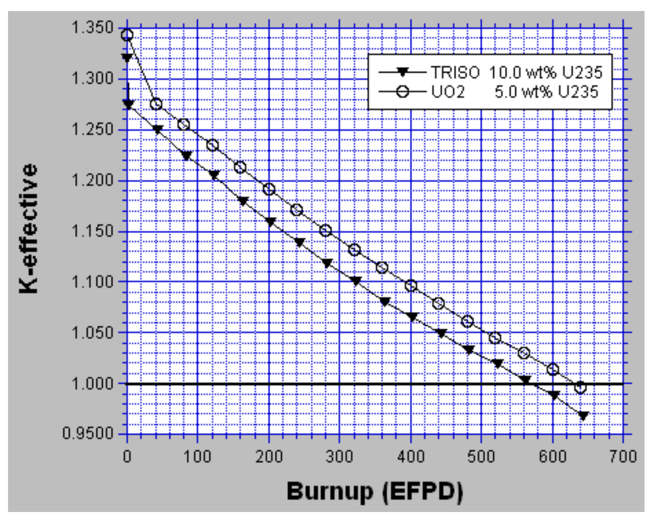

Figure 12. Core k-effective versus burnup for the TRISO and UO_2 loaded with 554 g U235 fuel blocks.

The second depletion calculation was for a uniform core load of reload blocks (776 g U-235 per block). For the

TRISO-coated particle fuel case, the enrichment was 14.0 wt% U-235 with a particle packing fraction of 0.24715, UCO kernel size of 425 μm in diameter, UCO density of 10.50 g/cm^3, and a fuel rod diameter of 12.45-mm. For the UO_2 case, the enrichment was again 5.0 wt% with a fuel rod diameter of 3.63-mm. The letdown curves (Figure 13) for the TRISO fueled core goes subcritical after approximately 890 EFPDs and the UO_2 core after 815 EFPDs. The UO_2 core EFPDs can be increased by increasing the uranium enrichment slightly.

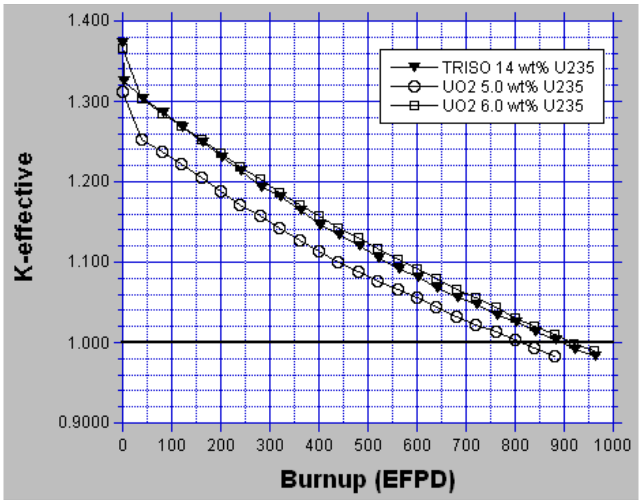

Figure 13. Core k-effective versus burnup for the TRISO and UO_2 loaded with 776 g U235 fuel blocks.

The third depletion calculation is essentially identical to the second depletion calculation, except the UO_2 core uranium enrichment was increased from 5.0 to 6.0. wt%. The fuel rod diameter had to be decreased slightly from 3.63 to 3.31-mm in order maintain the 776 g U-235 loading per fuel block. The increased enrichment was needed to increase the number of EFPDs in order to better match the calculated TRISO-coated particle fuel core burnup. The UO_2 core can now achieve 915 EFPDs (Figure 13). For a one percent increase in the UO_2 enrichment, the burnup is increased by 100 EFPDs and is now longer than the TRISO-coated particle fuel core by 25 EFPDs. Therefore, a UO_2 enrichment of slightly less than 6.0 wt% would be needed to match the TRISO VHTR core burnup for this core configuration.

VI. CONCLUSIONS

From a neutronic standpoint, the proposed new fuel/clad design concept can offer a substantial increase in core reactivity for the prismatic VHTR over the current TRISO-fueled core. Substitution of the TRISO fuel compact fuel rods for small diameter UO_2 fuel rods with a dedicated clad can improve both the neutron moderation and U238 self-shielding to achieve this reactivity gain. The reactivity gain can then be used to reduce the uranium enrichment from 10-20 wt% for the current TRISO core down to 4-6 wt%.

The proposed small diameter UO_2 fuel pellets or rods (2-4 mm outer diameter) should be able to be fabricated readily using known and previously used techniques. The widely used UO_2 fuel form and its extensive light water reactor fuel performance database should allow for reasonable prediction extrapolation to VHTR fuel temperatures. The high melting point will increase the thermal margin relative to TRISO fuel.

The high-temperature clad needed for the new concept, however, does not yet exist and will require a substantial developmental effort. Recent carbide clad development does offer future potential for VHTR application. Most notable is the silicon carbide clad development effort underway for light water reactor applications as a replacement to the current zircaloy clad fuel rod in commercial reactors. Preliminary test results and computational calculations show early promise for this application.

Although the relatively high VHTR fuel and gas coolant temperatures will be a challenge for the high-temperature clad development, a couple of benefits may ease the difficulty. First, the VHTR is a thermal spectrum reactor with a relatively low power density, hence the radiation damage rate will be lower than a light water reactor. Second, in the new concept here, relatively thick clads should be possible relative to the small fuel rod diameters which in turn should allow innovative clad design options such as multiple sheaths systems with specific performance attributes. Third, shorter length fuel rods, or rodlets, stacked vertically in the block fuel channels may also serve to reduce the UO_2 and clad fabrication lengths.

Based on the neutronic analysis, it appears that the new fuel/clad concept in the VHTR core would behave in a similar manner as the TRISO-fueled VHTR core. The Doppler fuel and moderator coefficients are similar, as are the thermal neutron spectra. Depletion studies show that the Generation IV goal of an 18-month power cycle can be met with 5 wt% enriched UO_2 in the initial core.

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